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Abstract

FUEL CYCLE ECONOMICS OF FAST REACTORS

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The fuel cycle for all nuclear power reactors represents a distinct phase for economic analysis. Fast reactors have the potential of significant breeding gain. In addition, they are less inhibited by choice of core structural material, and the isotopic composition of the fissile specii is of lesser importance than in a thermal spectrum. These represent somewhat difference parameters to the fuel cycle economic analysis for fast reactors than exist for thermal reactors.

This paper presents several fast reactor fuel cycles to point out the more critical aspects of their economics and the importance of some variables. The fuel cycle properties were determined by a computer code capable of making space-dependent fuel cycle calculations employing a 16-group neutron energy structure.

Three reference cases, each representing a "near future" reactor system of about 500 Mwe, are chosen, using carbide, metal or oxide fuel. These reference reactors are heterogeneous, sodium cooled systems using the uranium-plutonium cycle. The three reactors have core volume fractions of about 25% fuel, 61% coolant and 14% structure with carbide fuel, 22% fuel, 62% coolant and 16% structure with metallic fuel, and 32% fuel, 54% coolant and 14% structure with oxide fuel. The equilibrium physical properties such as isotopic composition of the cores and blankets relative breeding ratios, blanket to core residence times and the fuel cycle costs are given.

A series of studies having changes from the references are presented. For example, increasing power by an increase in reactor volume at constant power density decreases breeding ratio. However, the investment in fissile

material decreases per unit of power as size increases. This yields decreased fuel cycle costs which can be significant when consideration is given to improved fabricating and processing costs resulting from greater volume of material handled.

A decrease in specific power obviously results in larger inventory cost. Compensations can be realized in that larger diameter fuel pins or higher operating temperatures can be used. The former has lower unit fabricating costs, the latter higher system thermal efficiency. The cost variations of fabricating and processing are illustrated as well as the influence of type of processing.

While stainless steel is the most commonly considered core structural material, others may be desired or required for reasons of higher system thermal efficiency or compatibility with the fuel. Many of these probable structural materials have a neutronically deleterious effect increasing essentially in the order titanium, iron, chromium, nickel, vanadium, niobium, and molybdenum.

The characteristics of a comparable uranium-233, thorium fast power breeder equilibrium core and blanket fuel cycle is shown. The fuel cycle costs for this reactor are shown as well as some of the physics which leads to differences, such as the reduced breeding ratio. A design having a good sodium void reactivity characteristic used U-233, U-238 as core fuel, with essentially full plutonium removal when the fuel is recycled. The "equilibrium" fuel cycle data are shown for this concept.

A cost normalization is offered for the four 1000 Mwe fast reactor studies completed for the United States Atomic Energy Commission in late January 1964. These studies covered two carbide and two oxide fueled cores of distinctly different core designs.

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Abstract

ADVANCES IN AQUEOUS PROCESSING OF POWER REACTOR FUELS

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Processes for the recovery of fissionable and fertile materials from irradiated fuels by aqueous dissolution and organic solvent extraction and/or ion exchange have been extended for application to most types of power reactor fuels by the development of new fuel dissolution processes. Most research and development effort has been focused on the feed preparation steps, and the effect of fuel irradiation level on the solvent systems. New processes have been developed for the recovery of most radioactive materials, including fission products and the transuranic elements.

Fuel decladding and/or dissolution methods have been developed for the most widely used power fuels, tubular stainless steel or Zircaloy-clad oxides, as well as for the less important uranium-zirconium alloys clad in Zircaloy-2, UO_2--stainless steel cermets clad in stainless steel, and alloys of uranium and molybdenum.

Feed preparation for stainless steel clad--oxide core fuels can be accomplished by (1) shear-leach; (2) sulfex decladding; (3) electrolytic dissolution; and (4) dissolution in aqua regia. For Zircaloy-2 clad--oxide core fuels, shear-leach and the Zirflex are applicable. A mixture of nitric and hydrofluoric acids will dissolve both cladding and core.

The cermets of oxides and stainless steel require aqua regia (in titanium equipment) or electrolytic dissolution in nitric acid. Hydrofluoric acid or ammonium fluoride solutions containing HNO_3 or H_2O_2 will dissolve zirconium-uranium alloys.

Processing methods for carbide fuels include: hydrolysis, acid dissolution,

and burning, and for graphite-based fuels: burning, grind-leach, and disintegration. Plutonium/uranium oxides can be dissolved in nitric acid/hydrofluoric acid; and beryllium oxide-based fuels can be dissolved in hydrofluoric acid, sulfuric acid, or ammonium fluoride.

The solvent extraction systems (Purex of Thorex) can be modified to recover thorium, uranium, protactinium, and plutonium from the feeds prepared by the above methods. Tri-butyl-phosphate (TBP) is the most used organic solvent, although others may be equally good. Laboratory-scale tests with prototype power fuels irradiated to 50 000 Mwd/tonne indicate satisfactory solvent stability with TBP. New diluents and new solvent cleanup cycles have been developed.

Solvent extraction and ion exchange processes have been developed for neptunium recovery. Plutonium-242 and an americium-curium-rare-earth fraction have been recovered from completely "burned" plutonium-239 by TBP extraction and ion exchange. Processes for the recovery of the transuranium elements are being developed using tertiary amines, phosphonates, TBP, and alkyl phosphoric acids. Solvent extraction and ion exchange processes for the recovery and separation of all valuable fission products are being developed and tested. Basic research is providing a better understanding of chemistry, mechanisms, and applications. High capacity centrifugal contactors have been tested for plant scale use; other high throughput, low holdup contactors are being developed.

The important cost components of fuel recycle, of which reprocessing is a significant but not controlling part, have been estimated. These costs are discussed as a function of plant capacity.

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Abstract

FLOURIDE VOLATILITY PROCESSES FOR THE RECOVERY OF FISSIONABLE MATERIAL FROM IRRADIATED REACTOR FUELS

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In flouride volatility processes the fissile material in spent nuclear fuels is converted into volatile hexafluorides that can be separated from fission products and other materials by volatilization. Possible advantages are (1) fewer processing steps, (2) the use of reagents with low susceptibilities to deleterious radiation effects, (3) a uranium product (uranium hexafluoride) in the form desired for isotope separation, (4) radioactive waste products in solid form and small volume, and (5) a small shielded-space requirement. These processes are being developed for both high-and low-enrichment fuels.

In fused-salt processes for enriched fuels, the fuel is dissolved at 650C or less in a molten fluoride mixture by bubbling hydrogen fluoride through the melt, The uranium is then fluorinated to the hexafluoride at 500C and volatilized. The hexafluoride is decontaminated from fission products by a sorption-desorption cycle using sodium fluoride as the asborbant.

In a recent series of hot pilot-plant runs, highly decontaminated uranium hexafluoride was recovered. The highest individual decontamination factor (around 10^{10}) was for zirconium and the lowest was for ruthenium (around 10^{8}).

In the fluidized-bed fluoride volatility processes, the spent fuel material is submerged in a bed of fluidized inert alumina while dry gases are reacted with the fuel. The fluidized bed dissipates the heat of reaction. This process, when applied to enriched zirconium or aluminum fuels, has the following major steps:

(1) Hydrogen chloride is reacted with either zirconium (at about 400C) or aluminum. The zirconium or aluminum chlorides are converted to the oxides in a separate fluid bed by reacting with steam. (2) The uranium and fission-product chlorides remaining in the alumina bed are converted to the fluorides by hydrogen fluoride. (3) These fluorides are then fluorinated and the volatilized uranium hexafluoride is collected in cold traps. (4) Further decontamination can be achieved by distillation or by the procedures described above. This process is now being demonstrated in cold pilot-plant experiments.

Work is being done on the fluidized-bed fluoride volatility process for low enrichment fuels of uranium dioxide clad in Zircaloy or stainless steel. These fuels contain plutonium that must be recovered. The present process plan has the following major steps: (1) The uranium and the plutonium are separated from the fuel jacket. For zirconium, this separation has been done on an engineering scale with hydrogen chloride by formation and volatilization of the zirconium tetrachloride. A possible procedure for stainless steel cladding consists of removing the oxide from the chopped cladding through formation of powdered U_3O_8 by oxidation of the dioxide. (2) The oxides are fluorinated with fluorine gas at about 500C. (3) Distillation of the hexafluorides may be used to remove fission products and to partition the hexafluorides into uranium hexafluoride and a mixture more concentrated in plutonium. (4) The mixture of plutonium and uranium hexafluorides can be converted by reacting with hydrogen and steam to the oxides, which can then be fabricated back into reactor fuel.

The fluid-bed process has been partially demonstrated on an engineering scale using non-radioactive materials. Bench-scale studies show a potential for application to plutonium-bearing fuels. Facilities for pilot-scale demonstration using plutonium and tracer levels of fission products are nearing completion.

Additional versatility of the fluid-bed method for carrying out gas/solid reactions has been demonstrated by recovering uranium hexafluoride from graphite matrix fuels. At the present time a pilot plant is under construction to demonstrate this process for graphite-matrix fuels.

The technology of fluoride volatility processes is nearing the stage at which design can be started for a commercial-scale plant for enriched-uranium fuels. Additional development work is necessary to confirm the promising prospect for application of these schemes to low-enrichment power-reactor fuels containing plutonium.

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Abstract

PYROMETALLURGICAL AND PYROCHEMICAL FUEL PROCESSING

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High-temperature fuel reprocessing methods are being developed in order to achieve rapid recycle of discharged reactor fuel in as direct a manner as possible. With these processes, low fuel recycle costs are expected to result from minimum fuel inventory charges, avoidance or simplification of chemical conversions, small process volumes and, hence, compact processing equipment, direct production of solid wastes, absence of severe critical mass restrictions, and the possible retention of valuable fuel alloying additives. Because of these characteristics, pyrometallurgical processes can be employed for recovery of fast reactor fuels in processing plants located on the reactor site.

A processing facility adjacent to the Second Experimental Breeder Reactor (EBR-II) will be used to demonstrate various pyrometallurgical processes in connection with recovery of EBR-II fuels. The first EBR-II fuel, an enriched uranium-fissium alloy, will be recovered by melt refining, a simple melting procedure by which fission products are removed by volatilization and oxidative drossing reactions. Plant equipment for melt refining 10-kilograms batches of fuel has been installed in the EBR-II Fuel Cycle Facility and is expected to be in operation before the end of 1964.

Two other processes are required in the EBR-II fuel cycle to (1) recover U-235 from melt refining crucible residues and (2) concentrate the plutonium bred into the uranium blanket material. In these processes, molten metals (e.g., zinc and magnesium) and molten halide salts are used as solvents. Both processes have been advanced into the pilot plant stage of development. Other

processes utilizing liquid metal and molten salt media are being developed for the recovery of fast breeder fuels containing plutonium. Metal, oxide, and carbide fuels can be accommodated in the liquid metal-salt processes.

At Hanford, a closed-cycle process using a molten chloride solvent, such as a lithium chloride-potassium chloride system, is being developed in connection with the Plutonium Recycle Program (PRP). The fuel, a mixture of uranium dioxide and plutonium dioxide, is dissolved by sparaging the molten chloride bath with hydrogen chloride or chlorine. Recovery of the uranium and plutonium from the melt, in forms suited to recycle to the reactor, can be achieved by: (1) electrodeposition of essentially plutonium-free uranium dioxide, (2) electrodeposition of solid solutions of uranium and plutonium dioxides, and (3) precipitation of essentially uranium-free plutonium dioxide.

Reactions of gaseous reagents with uranium oxides and carbides are being used in fuel reprocessing studies at Atomics International. The objectives of the processes are limited to repair of radiation damage and re-enrichment of the fuel. Uranium oxide fuels are pulverized for refabrication by alternate oxidation with oxygen or air and reduction with hydrogen. One process for uranium monocarbide fuels consists of gaseous oxidation followed by carbothermic reduction of the oxide to regenerate the carbide. Another consists of gaseous nitridation to produce a uranium nitride-carbon mixture followed by reconversion to the monocarbide by heating the mixture in a vacuum to above 1500C. A third process, which, in contrast to the gaseous-reaction processes, provides significant removal of fission products, involves electrolytic dissolution of the carbide in a molten chloride bath. The metallic uranium product is dissolved in mercury and reconverted to the carbide by distillation of the mercury in a hydrocarbon atmosphere.

Present fission-product recovery and shipping programs at Hanford provide for annual capabilities of 5 megacuries of strontium-90, 1 megacurie of cesium-137, 5 megacuries of cerium-144, and small amounts of promethium-147 and technetium-99. Existing process systems are used for the recovery operations, final encapsulation and some purification steps are performed at Oak Ridge and Quehanna, Pa. Recovery processes are based on sulfate precipitation and D2EHPA extraction for strontium, cerium and the promethium-bearing rare earths and on alumino-silicate exchanging for cesium. Methods for cross-country shipping of megacurie amounts of fission products have been developed, and high safety standards have been established for control of the associated hazards. Strontium and cerium are shipped as precipitates in annular filter casks protected from impact, fire and immersion effects by heavy shielding and a special shock-absorbing structure.

Future production plans are aimed at encapsulating fission products at Hanford as a corollary to waste management. The long-lived, heat-producing strontium-90 and cesium-137 isotopes will be separated from the process wastes, and the residual solutions will be converted, after appropriate aging, to immobilized salt cakes in existing storage tanks. The strontium and cesium will be stored separately or made available for sale. Plans are being developed to establish an encapsulation plant at Hanford that will process and package in final form the strontium and cesium removed from the wastes; cerium, promethium, technetium and other isotopes may also be processed. Thus annual capabilities during the five year period 1968 - 1973 will be 10 megacuries of strontium-90, 10 megacuries of cesium-137, 100 megacuries of cerium-144, 30 megacuries of promethium-147, and kilogram quantities of technetium-99.

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Abstract

CONTROL-ROD MATERIALS

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In the papers on control-rod materials given at the 1958 Conference on the Peaceful Uses of Atomic Energy, the development of specific materials for specific applications was described, but the performance of materials in power reactors was not discussed. Since then, many control-rod materials have received thorough testing in power-reactor applications. Also, since 1958, the U. S. Atomic Energy Commission has sponsored a comprehensive program to provide data and develop technology on control-rod materials. Consequently, much information has been published since 1958 on control rod materials and their performance, and one book on the subject has become available.

The progress of control-rod materials technology in the United States is discussed; power reactor applications are emphasized. The materials discussed in some detail are: boron in alloys as compounds and in dispersions; hafnium in elemental form; the lanthanons as oxides, compounds, alloys, and dispersions; and the silver/indium/cadmium alloy.

Progress in fabricating boron-containing materials is discussed. Neutron irradiation effects and the extent of helium release are presented for boron-containing materials as related to boron burnup and crystal structure. Experience with boron in various forms is described for Dresden, the Vallecitos Boiling Water, Elk River, N. S. Savannah, Big Rock Point, Hombolt Bay, and SM-1 reactors. The low cost, simplicity of fabrication, and the excellent performance of vibratory-compacted boron carbide in tubes is discussed.

The use of hafnium, formerly confined to naval reactors, has been extended to commercial power reactors, Shippingport, Indian Point, second core of Yankee, and some blades in the Experimental Boiling Water Reactor. Hafnium availability

for commercial reactors is improved. Information on the performance of hafnium is still meager, but evidently hafnium has performed well.

The data on the lanthanons as alloys, oxides, and compounds are reviewed and their use in dispersions is described. The use of mixed gadolinium and samarium oxide in the Hallam reactor is discussed. A discussion of the fabrication problems in using europium oxide as a dispersion in stainless steel for the SM-1 and SM-1a reactors is given. Europium oxide has been utilized as a high-temperature control material for the HTRE by containing it in segmented nichrome cans. When used as a dispersion in stainless steel for the PM-1 reactor, europium oxide is formulated with titania to obtain the desired corrosion resistance. Newer compounds of the rare earths, such as the tungstates, are also discussed.

Control rods of nickel-plated, silver/indium/cadmium alloy were used originally in the Yankee reactor. During refueling of Yankee, dissolution of the alloy occurred where the nickel plating had worn, with subsequent deposition of radioactive silver on all reactor surfaces. The silver indium cadmium rods in Yankee were replaced with hafnium. Silver indium cadmium may still be used as the control material in the third core of Yankee, but clad with Inconel. The alloy also is in the Saxton experimental pressurized water reactor operating at 280C.

An economic evaluation of control materials is given in which the costs of materials and fabrication are balanced against lifetimes estimated from nuclear-worth depreciation.

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Abstract

DEVELOPMENTS IN SOLID MODERATOR MATERIALS

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Work conducted in the past several years at a number of laboratories in the United States has greatly extended the knowledge on solid moderator materials. Most effort has been devoted to three classes of materials: carbons and graphites, beryllium oxide, and metal hydrides. Some work has been reported on beryllium, although this material has not found extensive use as a moderator. In this review of solid moderators emphasis is placed on recent developments, potential problems, and especially on the radiation effects in these materials.

Graphite is used to greatest advantage in large power reactors or in applications in which the moderator is subjected to very high temperatures. Emphasis has, therefore, been placed on long-term irradiation studies in which the neutron dose approaches that expected during the lifetime of an advanced power reactor, on radiation effects at high temperatures, and on gas - graphite compatibility studies. Irradiation studies have now been extended to doses of 1.6×10^{22} nvt (E 0.18 MeV) at 800C. The most important effects observed relate to dimensional stability. Much more limited data have been obtained for irradiation temperatures up to 1200C.

Factors that affect the radiation-induced dimensional changes in graphite have been identified. For example, it has been determined that non-crystalline carbons contract at a much higher rate than the more crystalline graphites. These and other studies have led to a theory for radiation-induced dimensional changes that occur at high temperature. Based on this theory, new materials, which should have improved properties, were fabricated; irradiation tests are in progress.

The creep and mechanical properties of graphite have been studied under a variety of conditions, both in and out of reactor. It has been determined that an external stress significantly modifies the radiation-induced dimensional changes at high temperature.

Graphite is finding increasing use as a fuel matrix, fuel can, and coolant tube for high-temperature gas-cooled reactors, and potential problems in these applications are being explored. Compatibility of the graphite with trace amounts of oxidizing gases in helium-cooled reactors is a problem that will require careful purification of the gas.

Beryllium oxide has good strength at elevated temperatures, high thermal conductivity, and other favorable physical properties, and also is a good moderator with a relatively low absorption cross section. The major question in the use of beryllium oxide is its stability after extensive irradiation. A considerable amount of information has been obtained recently on this question. The range of conditions tested include neutron doses up to 2.8×10^{21} nvt (E 1 Mev.) and irradiation temperatures from 100 to 1200C. Earlier reports of powdering at low temperatures and high exposures have been confirmed; however, at high temperatures the dimensional stability is greatly improved. Radiation-induced expansion is attributed to three factors: crystallographic lattice expansion, grain-boundary separation, and helium bubble formation. It appears that reactor operating conditions may be chosen to control expansion from the first two effects, but a better understanding of the contribution to damage of helium formation is needed. Factors such as sample size, grain size, preferred orientation and purity have also been examined and were found to influence the irradiation behavior.

Zirconium and yttrium hydrides offer definite weight and volume advantages, and, therefore, find use in special moderator applications. However, the loss of hydrogen by thermal decomposition limits their use to relatively low temperatures, although the useful temperature range can be extended with cladding. The thermal stability of yttrium hydride is considerably better than zirconium hydride, and recent physical-property determinations have greatly enhanced its usefulness in engineering applications. However, more information on the radiation effects in both yttrium and zirconium hydrides is needed.

stainless steel in large fast reactors because of its low neutron absorption cross section. Limited alloy development and fabrication of specific vanadium alloys are discussed.

Recovery, recrystallization, and grain-growth data for zirconium and Zircaloy are discussed as a function of cold work, composition, annealing time, temperature, and annealing atmosphere. The influence of fabrication variables on mechanical properties and strain anisotropy in Zircaloy are presented.

Brief descriptions of the following are also provided: fabrication and testing of austenitic stainless steel cladding containing 300 parts/million boron; evaluation of stainless steel performance in the first Yankee Core; hot coestrusion of UO_2 and stainless steel; extrusion of ribbed beryllium cladding; coestrusion of Zircaloy/stainless steel tubing and tubular joints; pressure bonding; welding of spacer ribs to Zircaloy cladding; tungsten tube development for application in special reactors; improvements in Zircaloy tube fabrication; new zirconium alloys for water-reactor cladding; Zircaloy-clad elements swaged to reductions in area in excess of 40% and end-closure fabrication.

Paradoxically, this current relatively low state of uranium production and reserves coincides at this Conference with a renewed optimism on the early widespread uses of nuclear energy for civilian purposes. Production will continue to decline, but rapidly increasing uranium requirements are now predicted for the near future. The increasing requirements of the 1970's, following an extended period of decline in uranium production and exploration, may tax our abilities to find, develop and produce sufficient uranium.

Worldwide projections of requirements have not been developed in detail beyond 1980, but cumulative U. S. requirements may vary from 600 000 to one million tons or more by the year 2000, compared to current low-cost reserves of about 160 000 tons, and a possible, but uncertain, additional 400 000 tons in the western uranium mining areas which may exist in comparable deposits. Information on higher-cost material that might be available at prices up to $20 per pound is meager, but the quantities available may fall considerably short of doubling low-cost reserves.

Utilization of thorium in advanced converter and thermal breeder reactors may slow the drain on uranium resources when these systems start becoming operational. The U. S. is well endowed in thorium resources available at low cost. Advances in hydrometallurgical processes may delay the day when high-cost uranium or thorium is required or reduce the estimated cost of recovery from very low grade sources both directly and by economic recovery of by-products.

The geographical location of nuclear fuel resources is of little economic importance because of their high energy content. The availability of nuclear fuels may be therefore considered on a worldwide basis which should prolong their availability at reasonable costs.

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Abstract

APPLICATION OF GEOLOGIC CONCEPTS TO FUTURE URANIUM EXPLORATION

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Most of the present important uranium mining areas in the U.S.A. apparently have reached or passed maturity. Rapidly rising requirements for nuclear power forecast for the 1970's will greatly exceed expected additions to reserves from operating mines and nearby areas and will have to be met primarily by production from new areas yet to be discovered. The development and use of geologic criteria of uranium-ore genesis and habitat will be the dominant factors in determining both regional favorability and specific exploration targets; it is expected that the major portion of newly discovered reserves of low-cost uranium will be in sandstone-type deposits. Geophysical and geochemical methods will play an important role in detailed exploration of new areas, but it is less likely that they will be the initial means of locating them.

Geologic concepts conforming to empirical data on the occurrence of uranium in sandstones and conglomerates are presented which, it is believed, form a promising basis for evaluating the uranium potential of unexplored regions in the U.S.A. and elsewhere in the world. These presuppose a relationship between the derivation of uranium and tectonic events (which are also associated with the formation of hypogene deposits of various metals) but do not necessarily require that the uranium be hypogene.

From studies of metallic ore deposits on the American continents, it has become apparent that belts or centers of strongest intensity and highest-temperature mineralization in any region generally coincide with linear tectonic units or occur at intersections of such units or at volcanic centers. Mineralization occurs in zonal arrangement and paragenetic sequence along decreasing intensity gradients radiating from these focal areas. Although uranium can occur in all zones, it is most abundant in the outermost, shallow, low-temperature zones where it is deposited from solutions with telethermal or ground-water characteristics. Fluids responsible for mineral zoning may deposit hypogene uranium or they may remove uranium from granite, tuff or other rocks during alteration and chemically modify them sufficiently to enable more effective supergene leaching. Supergene leaching of fresh rock may also be triggered by tectonic events, and deposition may occur in the same marginal areas favorable for hypogene mineralization.

Within low-temperature zones, uranium deposition in sandstones is controlled by features peculiar to the sedimentary-rock environment. The character and distribution of unoxidized ore bodies in permeable carbonaceous sandstones indicate the importance of reducing and oxidizing processes in ore formation. Ore bodies are directly related to interfaces formed by the incursion of oxidizing ground waters into reducing zones created by anaerobic bacterial decay of organic matter. Potentially productive ground can be delineated by locating these boundaries by means of their conspicuous mineralogical features.

Areas providing coincidence of low-temperature hypogene zones, migration channels for supergene solutions and favorable sedimentary environments occur generally along the common margins of orogenic uplifts and adjacent continental basins. An assessment shows the uranium possibilities of various regions in the U.S.A. and elsewhere in the world where such favorable loci are presumed to exist.

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Abstract

REACTOR PHYSICS IN THE THERMAL ENERGY REGION

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The physical description of the thermalization and transport of neutrons below a few electron volts is very important in advanced thermal-reactor design. This description expressed in terms of low-energy scattering cross sections may be combined with the necessary fission, absorption, and fast transport cross-section data for calculation of reactor neutron spectra and performance. The many special numerical methods now available for accomplishing this task permit relatively great refinement to be incorporated in the parametric reactor design calculations providing the basic cross-section data are known. However, only in recent years has the treatment of low-energy neutron-scattering cross sections (thermalization) been considered in other than an elementary manner because of the complications introduced by moderator lattice structure and chemical binding.

Extensive work has been carried out in the U.S. to develop scattering kernels of the required accuracy for the common reactor moderators H_2O, D_2O, $(CH_2)_n$, C_6H_6, C, Be and ZrH_x. The procedure used has been to relate the theoretical scattering model to known physical and chemical properties of the moderator; i.e., temperature dependence of the specific heat, vibrational. and rotational frequency spectra, elastic constants, dispersion relations, total neutron cross sections, etc. Another procedure has been to use experimentally determined $S(\alpha,\beta)$ to derive a phonon frequency distribution. Use of $S(\alpha,\beta)$ directly in reactor calculations has been impractical because experimental data are not available over the entire range of variables α, β and temperature.

The adequacy and limitations of the theoretical scattering kernels have been studied experimentally for the above moderators by measurements of steady-state spectra, spatial flux distributions and time-dependent behavior under experimental conditions applicable to reactor analysis. Verification of the theoretical scattering kernels from the standpoint of the reactor analyst represents a fairly extensive undertaking since thermal spectra and other reactor parameters must be measured for theoretical comparisons over a wide range of temperature ($\lesssim$ 60K to $\gtrsim$ 1000K), absorption, and spatial fluxes. An investigation of neutron spectra in clean geometries approaching those of an infinite medium constitutes one of the simplest and most appropriate tests of thermal scattering kernels. These studies have been extensively carried out in the U.S. at ambient temperatures for the common reactor moderators. It has been shown that the scattering kernels based on a knowledge of the molecular or crystalline structure of the above moderators permit the reactor analyst to predict neutron spectra with far more precision than possible with the previous free-atom scattering descriptions or various arbitrary prescriptions. Many temperature-dependent spectral experiments have been carried out, and, again, the more realistic scattering descriptions are beginning to provide the accuracy required for present reactor analysis. In addition, these measurements, which essentially establish the validity of the inelastic isotropic component of the scattering kernel, serve as the basis for further model refinements and adjustments in the scattering models.

Angular flux spectra (spectra in finite media) have been especially difficult experimentally to study unambiguously. These measurements provide an important test of both the isotropic and the anisotropic components of the scattering kernel. Experimental results indicate some significant discrepancies in the theory under boundary conditions similar to those in actual reactor lattices. Other experimental investigations which serve to test thermal scattering kernels from the standpoint of reactor design are measurements of thermal diffusion length by pulsed or continuous absorption techniques, thermal flux distributions at spatial and temperature discontinuities, and time-dependent spectral studies.

Generally applicable numerical techniques for the calculation of thermal scattering kernels are being developed in the U. S. and tested by the above experimental procedures These flexible methods are planned to incorporate many special provisions required by users; for example, the choice of moderator, energy mesh, moderator temperature, and crystalline structure.

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Abstract

PHYSICS OF FAST REACTORS

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Methods used in the study of the physics of fast reactors are reviewed with emphasis on advances made in the U.S. since 1958. Areas considered are: microscopic cross sections; multigroup constants and methods; statics; sodium void coefficient; and Doppler effect.

Recent advances in the knowledge of nuclear cross sections pertaining to fast reactor applications are reviewed and summarized. Experimental results discussed include: a) the differential elastic and transport cross sections of a wide range of elements extending from Al to U; b) detailed inelastic excitation cross sections; c) fast neutron capture inclusive of fissile materials; and d) fission properties particularly $\bar{\nu}$ and $d\bar{\nu}/dE$. Theoretical models are employed to interpret and extend the experimental results. In particular, recent advances in the optical model of neutron scattering and the theory of neutron induced reactions are employed.

A new 26 energy group set of multigroup constants, based on the latest microscopic data, has been generated. The heavy element constants are composition independent except for the region below 40 keV where resonance self shielding effects are included. The light and medium element transport and elastic transfer cross sections are composition dependent and are obtained from a many group fundamental mode calculation. The composition independent values of the multigroup constants are tabulated for some key materials (U-235, U-238, Pu-239) as are some typical composition dependent multigroup constants.

A set of fast critical assemblies is selected so as to cover the range of compositions and sizes studied experimentally to date. Various diluents (U-238, Al, Fe) and fuels (U-235, U-233, Pu-239) are included. Comparison is made for this set between calculations, based on the latest microscopic cross section data, and experimental results.

The dependence of the various components of the sodium void coefficient of plutonium fueled reactors on reactor size and composition is studied. The positive spectral effect is considerably increased by the presence of Pu-240, fission products, and certain refractory metals. It appears possible to compute the spectral and capture components of the sodium coefficient with a fundamental mode calculation, for which many groups may be used. The leakage component can be obtained with fewer groups. Available comparisons of calculated and measured sodium reactivity effects are presented.

While considerable uncertainties in resonance parameters exist for fissile isotopes, it does not appear that serious errors in Doppler effect calculations for large fast reactors will be caused thereby. The main contribution to the Doppler effect comes from the fertile isotopes, for which the parameters are reasonably well known. Effects studied and not found to present serious problems are the following: multilevel rather than single level fitting for fissile materials; the non-Maxwellian distribution of nuclear velocities; the temperature profile across a fuel rod; and heterogeneity effects. An improved method, which allows for proper averaging over resonance parameter distributions, has been developed for calculating overlap effects in fissile isotopes and overlap between fertile and fissile isotopes. The improved method is accurate for energies down to several hundred eV.

The most successful recent improvements in basic reactor shield attenuation calculations are principally associated with the Monte Carlo method, for which the techniques of importance sampling, generalized quota sampling, and the use of expected values have been further developed. Another technique, numerical integration of the Boltzmann equation (the NIOBE code), has been very useful in the calculation of neutron spectra in shields, although with present available computer capacities it is not applicable for source distributions in which the forward-scattered component can predominate. For situations in which the geometry is complicated it has been expedient to integrate point-to-point attenuation kernels.

Shielding experiments have advanced principally in the detail of measurement. Thus neutron spectra in a water shield around a reactor have been obtained with a semiconductor spectrometer. The transmission of radiation through concrete has been measured for the case of a collimated source at several angles of incidence. And clean measurements have been made of the secondary gamma-ray production in a variety of shield materials surrounding a small neutron source. These last measurements have been used to provide input data for the shield-synthesis technique, in which shields are designed systematically by choosing materials or mixtures of materials for optimum performance by consideration of their density, neutron and gamma-ray attenuation, and secondary gamma-ray production.

Recognizing the importance of keeping track of shielding information, the U. S. Atomic Energy Commission has formed at Oak Ridge National Laboratory a Radiation Shielding Information Center. This organization collects and evaluates shielding information, records an evaluated list of sources on computer tape, and publishes a readout (which is an annotated bibliography) from time to time. It also collects shielding computer codes and investigates their relative capabilities. It assists shielders in finding information or obtaining codes on request.

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Abstract

DEVELOPMENTS IN TRANSPORT THEORY

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Recent efforts to broaden the theoretical base of neutron and photon transport include (1) systematic derivation of a transport equation from the formalism of quantum field theory using coarse-grained density distributions; (2) use of probability-generating functions together with first-collision equations to formulate a stochastic theory of neutron transport, including arbitrary deviations from the mean; and (3) invariant imbedding methods wherein the usual Boltzmann or transport equations with conditions at two boundaries are replaced by non-linear equations with conditions at a single boundary.

Recent achievements in the classical analysis of the transport equation and related linear integral equations are considered. The properties of the spectrum of eigenvalues of the transport operator are reviewed, and the method of expanding general solutions in terms of a complete set of eigenfuctions is outlined. Extensions of the eigenfunction expansion method from plane to curved geometries, from isotropic to more general scattering laws, and from monoenergetic to energy-dependent problems are sketched. Limitations and potential extensions of the method are noted.

Recent developments in integral theory and variational methods are summarized, including methods yielding a result with an error of high order in the trial function error. The analytic section of the paper concludes with a review of some

of the useful transformations and simplifications of the transport equation, in particular of its source term.

The second section of the paper is concerned with numerical methods. The generalized discrete-ordinates multi-group method is first discussed beginning with formulation of integral equations, called mesh cell balance equations. Various simplifications, in particular diffusion theory, are considered in this framework.

Next, the application of difference schemes to the balance equation is discussed, and then the rules for evaluating the resulting recursion formulas are described. Here the major concern are the control of errors, the handling of implicit source terms and boundary conditions, and the convergence of multi-level iterative processes. The matter of inner, outer, and eigenvalue iterations and associated convergence test is examined briefly.

Methods that minimize or dispense with inner iterations are noted in connection with the diffusion equation. Here the reflection method, related to the concept of invariant imbedding, is extended to multi-dimensional problems. Advances in the techniques for solving the second-order form of the diffusion equation and the extension of these techniques to the transport case are noted. Some numerical comparisons of various solution methods are presented.

Finally, progress in non-discrete numerical techniques, in Monte Carlo procedures, and in combination of the latter with deterministic methods is summarized.

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Abstract

DEVELOPMENTS IN RESONANCE ABSORPTION

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Analytical methods for calculating the resonance absorption of neutrons in reactors have been greatly improved since 1958. Improvements include Rothenstein's extensive work on second-order corrections (1959), the Goldstein and Cohen intermediate resonance formulas (1962), and Nordheim's numerical code for the direct solution of an energy-dependent integral equation (1961).

The close analogy between the integral equations for homogeneous and heterogeneous systems has led to a series of equivalence theorems of generally increasing sharpness. These include Bakshi (1957), Dresner (1958), Bell (1958), Rothenstein (1959), etc. In tightly packed lattices the Dancoff correction is important. Improvements in its determination for rod lattices were made by Thie (1958), Takahashi (1960), Fukai (1960) and Sauer (1962). Levine (1963) has found improved equivalence relations for a variety of uranium-bearing rods in water and of uranium-metal rods in D_2O and graphite.

As cognizance of the accuracy of the theoretical methods has spread, more accurate integral experiments have been devised to test aspects of the theory. In particular, the work of Hellstrand and his collaborators, of Hardy, Klein and Smith of Bettis and of Kouts' group at Brookhaven have resulted in sharp improvements in the precision of these integral experiments. Examples are resonance integrals measured by Hellstrand, et al (1960, 1961) and by Rothman and Ward (1961), and the Bettis measurements of Dancoff factors in water lattices (1962) and of the spatial distribution of resonance capture in a uranium rod (1961). Considerable improvement has also been obtained in experimental measurements of Doppler coefficients. The importance of Doppler

coefficients to the safety of large fast reactors has spurred considerable interest in their measurement and in obtaining improved neutron resonance data in the statistical region.

The present generation of very high speed digital computers has broadened the use of Monte Carlo calculations. Many of the limitations of the original REP code have been eased, including limitations on the number of resonance absorbers and on geometrical routines. A remaining difficulty is that a proper representation of multilevel interference effects in closely spaced resonances of fissionable materials is still not available. Some progress is expected from the present concerted effort on measurements of low-energy resonance structure and from better methods of curve fitting such as that proposed by F. Adler. The Monte Carlo code MOCCA perfected by H. Rief has made it possible to investigate high-energy resonance capture with good accuracy. Studies by Levine (1963) have clarified the differences between resonance capture in a 1/E spectrum and those of an actual reactor lattice.

Resonance absorption in water lattices has been thoroughly studied by Drawbaugh (1960), Rothenstein (1960) and Levine (1963). The results of these calculations have helped improve the accuracy with which the nuclear properties of these lattices can be predicted, as shown by Hellens and Honeck (1962). Similar studies are proceeding for other moderators. The adequacy of simple methods of dealing with rod clusters is still largely untested.

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Abstract

ADVANCEMENTS IN ANALYTICAL METHODS AND MACHINE COMPUTATIONS IN REACTOR PHYSICS DESIGN

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The continued improvement in reactor design is evidence of the advances made in techniques of reactor-physics analysis. These advances have been stimulated by increased knowledge of microscopic cross section data, improved numerical methods, and the availability of digital computers with more versatile capabilities. It is the purpose of this paper to review these developments and, where possible, to assess their adequacy for current design purposes.

Few-group diffusion theory has retained a dominating role in the analysis of reactors because of (1) its intrinsic simplicity, (2) wide applicability for calculating multi-dimensional neutron flux distributions, and (3) the highly developed technology for numerical solution of the group diffusion equations. With regard to the latter, there have been improvements in the now standard interative methods used for solving multi-dimensional problems. These include, for example, variational derivations of difference equations for non-orthogonal mesh coordinate systems, application of multi-line successive overrelaxation and improved alternating-direction implicit-iteration techniques, and refinements in inner-outer iteration strategy. New non-conventional concepts are being investigated for extending the scope and efficiency of group diffusion computations.

However, diffusion theory remains an approximation to the more rigorous but computationally cumbersome transport theory. The need for better approximations has stimulated the development of auxiliary artifices for gaining accuracy in treating spectral and transport effects without leaving the

simplicity of the diffusion-theory framework. More accurate representations of the thermal-energy neutron spectra, based upon detailed scattering kernels and allowing for spatial spectrum dependence, have been developed. One very useful approach involves application of variational methods to derive "overlapping thermal group" constants. Similar devices, based on variational methods, have been used to introduce other transport effects into the diffusion framework. Another major diffusion-theory limitation, the problem of homogenization both in space and energy, has received continued attention. Methods for calculating spectrum (e.g., resonances) and transport effects in relatively simple geometries have been improved.

The calculation of the lifetime behavior of a reactor is greatly complicated by the presence of control rods, voids, xenon, fission products, and other time-dependent quantities. Improved methods have been developed for "synthesizing" one- and two-dimensional distributions to obtain accurate three-dimensional distributions. These provide detailed spatial representations of complex configurations for which neutron fluxes can be readily computed. These models may be easily coupled with highly automated programs that yield group constants as a function of depletion, allowing for spectrum and self-shielding changes.

In the development of the diffusion-theory formalism discussed above, a continuing effort has been made to balance requirements for accuracy and details with the need to keep the calculational method simple and fast enough for design purposes. To assess the validity of these methods and to replace them where necessary, other techniques that essentially neglect this balance and incorporate the maximum attainable accuracy and detail must be developed. For this reason, there has been increased emphasis in numerical methods for solving higher-order approximations to the transport equation in one and higher dimensions. These include the discrete ordinates, spherical harmonics, and Monte Carlo methods, which are becoming more prevalent among the programs for evaluating transport, slowing-down, and thermal spectrum effects.

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Abstract

LATTICE STUDIES AND CRITICAL EXPERIMENTS IN LIGHT WATER MODERATED SYSTEMS

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This paper reviews the status of reactor physics relating to slightly enriched uranium-water moderated reactor systems. An evaluation is made of the theories and experiments which have been reported by various laboratories in the United States. For purposes of simplicity the paper is limited primarily to heterogeneous reactor systems containing but a single region.

Published reports from the following laboratories have been considered for review: Argonne National Laboratory, Babcock and Wilcox, Bettis Atomic Power Laboratory (Westinghouse), Brookhaven National Laboratory, General Electric (Vallecitos), General Electric (Hanford), and Westinghouse Atomic Power Division.

In general, the paper deals with the measurement of reactor parameters and comparisons of the experimental results with theoretical calculations. Some of the following parameters are considered for review: buckling, eigenvalues, resonance capture, fast fission, thermal neutron distributions, thermal neutron spectrum, and dk/dB^2. In particular, the development of both theoretical and experimental techniques since the Second Geneva Conference (1958) is discussed. Certain experimental techniques such as the small lattice experiments, the variable loading method in exponential assemblies, and resonance capture techniques are reviewed. In addition, some references are given to work at B & W related to lattices containing U-235 and thorium, and to work at BNL on lattices containing U-233 and thorium.

Some typical results are given within the text of the paper while references to the complete works are listed in the bibliography.

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Abstract

FAST CRITICAL EXPERIMENTS AND THEIR ANALYSIS

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Fast-reactor critical experiments in the United States since 1958 have been conducted by three principal organizations, the Argonne National Laboratory, the Los Alamos Scientific Laboratory and Atomics International. The experimental results obtained and their analysis, although strongly overlapping, are primarily concerned with systems of different types. A brief description of each facility will be given and the experimental results and analysis discussed.

Argonne National Laboratory's Zero Power Reactor No. III (ZPR-III) located at the National Reactor Testing Station in Idaho has conducted studies on 23 different assemblies in addition to those reported at the Second Geneva Conference. In contrast to those previously presented, which were, in the main studies of metal-fueled systems including a few with graphite added for spectrum softening, this report will present results on metal, oxide, carbide, and cermet fueled reactor assemblies ranging in size up to 665 liters, and with critical masses up to 580 kg of uranium-235. Included in these are two studies on possible meltdown configurations. Also included are studies of two zoned loadings, (1) where the central zone simulates the composition and neutron spectrum of a previously built reactor, to permit comparisons of zoned vs. full critical techniques, and (2) where the central zone simulates the composition and spectrum of a 5000-liter uranium carbide fueled core. On this latter core, measurements of the sodium reactivity coefficient and the Doppler effect for uranium-238 will be reported. Other measurements to be reported

include critical mass, reactivity effects of various materials, spectrum determinations using fission ratios of U-235, U-238, U-234, U-236, Pu-239, Pu-240, and Rossi-α.

Two plutonium-fueled assemblies have been constructed and will also be discussed. One was a subcritical assembly, built to study the effects of self-heating and spontaneous fission in plutonium. The second was a mock-up of the French Rapsodie reactor.

Extensive analysis of these systems indicates agreement between the calculated and experimental critical size for a wide range of compositions using a modified Yiftah, Okrent, Moldauer cross-section set. Calculation of danger coefficients, fission ratios and neutron lifetime exhibit discrepancies. These results will be discussed.

Argonne's Zero Power Reactor No. VI (ZPR-VI) located at Argonne, Illinois, first started operation in July, 1963. Two U-235 fueled assemblies have been constructed. The first was a repeat of ZPR-III Assembly 22. The second was a 650-liter uranium carbide fueled reactor simulation. Emphasis has been placed on measurements of the reactivity coefficient of sodium density. These results will be discussed.

Atomics International's Advanced Epithermal Thorium Reactor (AETR) critical facility has been used to study 10 assemblies in zoned core loadings. Test zone sections with U-233 and U-235 fuel have been constructed with median fission energies varying from 422 keV to 0.0085 keV. In addition to fission ratio and danger-coefficient determinations, a measure of neutron importance as a function of energy was obtained by measuring the reactivity effect of introducing a series of carefully calibrated sources emitting neutrons of different energies. The Doppler effect for Th-232, U-235, and U-238 has been measured in an assembly with a median fission energy of about 190 keV. The results of the experiments and a comparison with calculation will be given.

At the Los Alamos Scientific Laboratory fast reactor studies have included experiments with the Los Alamos Molten Plutonium Reactor Experiment (LAMPRE). Determinations of the critical mass as a function of temperature and control reflector position have been made. Experimental results and their analysis will be reported.

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CALCULATION OF CRITICALITY, POWER DISTRIBUTION AND FUEL BURNUP IN THERMAL POWER REACTORS

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Analytical methods currently in use for the calculation of criticality, power distribution and fuel burnup in thermal power reactors are reviewed. Emphasis is placed on specific phenomena to be accounted for in the calculations and on procedures for integrating the basic elements of the calculation into an overall analysis. Aspects of the analysis are discussed in terms of their significance for specific reactor types.

Elements of the criticality calculation are examined with emphasis on the analytical methods being utilized in the treatment of neutrons slowing down and for the determination of reaction rates as a function of energy in the reactor core. The status of the nuclear data for important fuel and structural materials is discussed in terms of the correlation of calculations with critical experiments and of the effects of uncertainties on specific aspects of design calculations. Alternative procedures for the homogenization of the fuel-rod moderator lattice unit cell are described and compared. In the energy range of the fission source, fast fission and inelastic scattering, analytical methods involving lattice escape probabilities are compared with those based on Monte Carlo calculations. In the epithermal energy range, methods of analysis based on the use of resonance parameters in Monte Carlo calculations or in calculations as developed by Chernick and Nordheim are compared with methods based on the utilization of experimental resonance-integral data. In the thermal-energy range, alternative procedures for the determination of the unit-cell disadvantage factor are described, and the results of calculations utilizing various approximations are inter-compared.

Significant factors in the determination of the overall reactor power distribution are described. Complications arising as a result of the geometrical complexity associated with non-uniform fuel loading and partial re-fueling schemes are pointed out. Methods for the determination of group constants for few-group diffusion theory are described, and comparisons between calculated and measured power shapes in complex reactor configurations are presented. Effects of nuclear/thermal/hydraulic interactions on the power distributions in water-moderated reactors are considered, and methods for accounting for such effects are described. Problems arising from the sensitivity of the power shape to local reactivity in low-buckling cores are discussed. Limitations of earlier methods for the determination of three-dimensional power shapes are discussed, and the advantages associated with modern synthesis procedures for the construction of three dimensional power shapes are described. Data are presented comparing the results of conventional numerical calculations of neutron-flux distribution with those obtained utilizing flux-synthesis techniques. Relative computing times are also discussed.

Calculational methods utilized in reactor nuclear design for the determination of reactivity lifetime and for the specification of fuel-cycling procedures are reviewed. The influence of the reactor moderator properties and of the variation with lifetime of the characteristics associated with reactivity control on the analysis of reactor fuel depletion is considered. Special characteristics of the thorium/uranium fuel cycle and of the uranium/plutonium fuel cycle are indicated. Experimental data available for the evaluation of reactor fuel burnup calculations are reviewed, and comparisons between the results of fuel-burnup calculations and spent-fuel isotopic analysis are presented. Important processes involved in reactor fuel depletion are discussed, and the limitations associated with the methods of analysis currently employed are outlined. The potential of new methods of analysis, based on the calculus of variations, for the study of reactor power distribution as a function of time is discussed.

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Abstract

NUCLEAR CONSIDERATIONS IN THE DESIGN OF LARGE FAST POWER REACTORS

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The large, 1000 megawatt (electric), plutonium-fueled, fast power reactors under study today exhibit major differences in their nuclear characteristics from the U-235-fueled reactors of modest size currently in operation. Smaller reactors with medium enrichment fuel possess a harder neutron spectrum, a higher potential breeding ratio, and a considerable flexibility as to acceptable structural materials. Both the Fermi and EBR-II reactors employ an unrestrained, one-piece, metallic fuel element, which should expand axially during a power excursion, losing considerable reactivity and helping to terminate the accident. Both reactors have very small Doppler effects and substantially negative sodium void reactivity coefficients.

With the shift in attention to large core sizes and to ceramic or restrained metallic fuel elements, there has been a considerable reduction in the magnitude and reliability of the fuel expansion reactivity coefficient. Fortunately, the likelihood of a large, negative Doppler coefficient, has arisen. However, the prospect of large positive sodium void reactivity effects, overall or locally, has also entered.

In optimizing the nuclear design of large fast power reactors, one faces major conflicts among the requirements introduced by economics, by breeding, and by safety. The thicker blankets associated with higher breeding ratios can penalize economics. Cladding materials like niobium or molybdenum are harmful to breeding and to reactivity coefficients. Deliberate spectrum softening enhances the Doppler effect, but damages breeding.

The first section of this paper deals with the neutron physics performance of a wide range of large fast power reactors, using a newly developed cross section set. Critical mass, breeding ratio, and reactivity coefficients are

presented as a function of reactor size and composition, which in turn are related to reactor power. The effect of various structural materials on performance is examined, as are the differences between "new" and "equilibrium" reactors, and the influence of various blanket design modifications. The effect on performance prediction of various degrees of refinement in the treatment of resonance elastic scattering is discussed. Results obtained with the new cross section set are compared with those of previous sets. Fast reactor performance with the thorium-U-233 fuel system and with hybrid fuel systems is also treated.

The second part of the paper deals with the relation between reactivity coefficients and reactor safety. Efforts to set criteria for Doppler coefficients and sodium void effects are reviewed. Estimates of potential sodium loss rates are made and translated into reactivity insertion rates as a function of reactor design. The relation between Doppler coefficient, sodium void effects, fuel element type, power density and core damage is discussed for some typical accidents.

The third portion of the paper surveys the nuclear designs reported in the four independent 1000 Mwe studies for fast power reactors submitted by various American industrial groups to the United States Atomic Energy Commission in January, 1964. The separate design approaches have led to pancake-shaped reactors, to a large annular core, and to a cluster of more highly enriched, cylindrical cores of modest diameter. The salient nuclear features of these widely differing design approaches are reviewed herein.

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LATTICE STUDIES AND CRITICAL EXPERIMENTS IN D_2O MODERATED SYSTEMS

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Critical and exponential experiments at Brookhaven, Hanford, Massachusetts Institute of Technology, Savannah River, and the Westinghouse Laboratories have, over the past five years, provided a very firm experimental basis to the reactor physics of uranium-D_2O systems. The major emphasis has been on buckling measurements where over 200 different lattices have been studied covering a wide range of lattice types and fuel shapes. Activation techniques have also been developed to measure such quantities as fast-to-slow-fission ratios, resonance neutron captures, conversion ratios, and neutron spectral distributions. However, the activation measurements have been applied to a smaller range of lattices than the buckling measurements. The most recent studies have been particularly concerned with the problems of heterogeneous loadings and of anisotropic effects.

The critical and exponential data have been supplemented by substitution studies both of the conventional type and of the null-reactivity variety as practiced at Hanford and United Nuclear. The substitution measurements are particularly useful in providing differentials or in making lattice surveys where only limited fuel samples are available. An interesting new addition to the substitution facilities is the Plutonium Recycle Critical Facility (PRCF) for studying irradiated fuel at Hanford. Pulsed neutron techniques in D_2O systems have found their main application in studies of D_2O itself, but are now being applied to lattices as well, particularly in the university programs at MIT, North Carolina State, and the University of Florida. A unique experimental facility is the SPERT-II reactor, which is being used for shutdown and transient studies on D_2O lattices.

Although most of the U.S. work to date has concentrated on uranium systems, limited work is now underway on plutonium and thorium systems. The plutonium-bearing lattices are being studied to provide information on plutonium buildup effects in natural uranium reactors, on plutonium recycle systems, and on plutonium irradiations to produce transuranic elements. The D_2O - thorium systems are being studied both for thorium breeder reactors and for the production of U-233.

With this wealth of experimental information at hand, many of the calculational codes are primarily recipes designed for interpolation between the experimental points. A number of calculational methods have been developed for the detailed examination of specific aspects of D_2O reactor physics. Examples are various Monte Carlo codes for studying neutron slowing-down processes and resonance capture, the THERMOS code developed by Honeck for examining thermal neutron spectral effects, and the HERESY code developed by Klahr for examining heterogeneous effects. Good results have also been obtained, particularly by Westinghouse with their CANDLE and PDQ codes, in adapting light water calculation methods to the D_2O reactors.

As the D_2O lattice data are increasingly translated into actual reactor designs, the physics information has been combined with engineering and economic information in design and cost optimization calculations, such as Savannah River's FAD code. There has also been heavy emphasis on the operational aspects of lattice design in the evaluation of control systems, of flux shaping capabilities, and of the reactor operating coefficients. A number of special codes, such as OX and LASS, have been developed to combine these data for stability studies. Most of the experimental information on operating coefficients has come from the zero-power experiments; however, significant contributions are now being obtained from the various D_2O reactors operating in the U.S. including the Savannah River production reactors, the Carolinas Virginia Test Reactor, the Heavy Water Components Test Reactor, and Plutonium Recycle Test Reactor, and the research reactors, such as CP-5 and the MITR.

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FISSION PRODUCT BUILD-UP AND LONG TERM REACTIVITY EFFECTS

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The complex of reactors at the Test Reactor Area (TRA), National Reactor Testing Station in Idaho, provide a unique combination for the measurement of long-term effects in samples irradiated to large nvt values. The Materials Testing Reactor (MTR) and the Engineering Test Reactor (ETR) constitute irradiation facilities in which flux levels approaching 10^{15} and a considerable range of spectra are available, while the Advanced Reactivity Measurement Facilities I and II (ARMF-I and ARMF-II) constitute measuring instruments capable of precision of $\leq \pm 2.0 \times 10^{-7}$ $\Delta k/k$ in reactivity changes. These latter two pool-type reactors are improved versions of the original Reactivity Measurement Facility (RMF) described at the Second International Conference on the Peaceful Uses of Atomic Energy. The application of this type of facility to the subject problems was demonstrated in the 1958 paper. However, the scope has been broadened by the use of two reactors having similar operating characteristics, with different responses to thermal and resonance absorbers, but retaining radial symmetry about the sample axis.

These reactors have been used to study the effects of long-term irradiations, in a variety of spectra and at various neutron flux levels, on fissile, fertile, and neutron absorbing materials. These studies, using small samples, have been aimed at determining how k-infinity, conversion ratio, and the neutron absorption properties of control materials vary as a function of the length of operation of various full scale power reactors, in the thermal and near thermal neutron energy range. Measurements of fissile materials include the irradiation of:

a) test fuel plates directed toward the study of compositions and fabrication methods, b) special samples for the measurement of I-135 yields in fission, c) gross fission product poison studies, and d) specific examples of proposed reactor core cell configurations. The fertile material studies include the U-238-Pu-239 and Th-232-U-233 irradiation chains, and the isotopic distribution of the products. The third area includes studies on the irradiation of such multiple-isotope poisons as dysprosium, hafnium, and europium.

Each of these studies, because of conditions peculiar to their nature, requires a variety of considerations dealing with sample design, methods of transfer between irradiation and measurement facility, heat transfer, and data interpretation. The paper includes a progress report on investigations now underway.

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POINT REACTOR KINETICS AND STABILITY CRITERIA

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The purposes of this paper are: (a) to examine the form of point reactor kinetics equations that might be used for the correlation of severe transient experimental data; (b) to review existing stability criteria and to discuss some practical aspects of nonlinear reactor stability.

To this end, the point kinetics equations derived by different authors from fundamental transport theory are reviewed and reinterpreted. It is observed that these equations include terms that depend on the rate of change of the shape of the neutron density distribution throughout the reactor core. This rate of change may become appreciable even when the shape changes are relatively small but occur over a short period of time. It is concluded that shape changes should be considered when correlating severe transient experimental data. In support of this conclusion, some numerical results are presented.

Other conclusions that may be derived from the reinterpretation of the kinetics equations are:

a. Reactivity is an ultimately bounded function as it should be since there is only a limited number of neutrons per fission.

b. Conclusion (b) suggests that no reactor can admit a finite escape time unless fuel is injected at an infinite rate.

c. Point kinetics equations can always be used to answer questions of ultimate stability or instability.

Next, stability criteria for nuclear reactors are discussed. It is observed that necessary and sufficient conditions for stability with respect to small perturbations are well established both theoretically and experimentally. It is also noted, however, that instability with respect to small perturbations does not necessarily constitute a safety hazard. Examples of bounded oscillations are given.

Regarding nonlinear stability, it is observed that most of the derived sufficient criteria for reactor stability pertain to global asymptotic stability and they are very conservative. It is suggested that less restrictive conditions for stability might be established by including in the analysis some practical limitations. Specifically:

a. Analytical approximations for feedback reactivity should be functions that are ultimately bounded.

b. The analysis should include delayed neutrons since stability requirements derived without delayed neutrons may be so conservative as to be of no practical importance.

c. Stability criteria should be derived for a limited range of operating power levels and/or reactivity inputs since no reactor can be built to withstand an infinite power or energy storage.

Admittedly, the inclusion of these practical aspects in the analysis renders the problem of stability more involved but the results are more meaningful. This important point is illustrated by means of some practical examples.

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Abstract

SPACE TIME REACTOR DYNAMICS

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In order to extract the heat of a nuclear reactor efficiently and safely the nuclear designer should be able to calculate the detailed spatial distribution of the power generation. In principle, he should be able to calculate this distribution for all times during the reactor life, for all static and transient operating conditions, and for various postulated accidental transients. In practice, calculational tools providing this ability have not been available; instead, in design calculations the reactor is usually represented by a one- or two-dimensional model and, for the calculation of transients, often by a single spatial point.

Such methods have sufficed thus far; however, there is growing interest in more precise treatments of spatial effects both in static and dynamic calculations. Partly this is a result of the trend toward large power reactors in which spatial effects become important. In large reactors power shapes tend to show large distortions in response to small disturbances. The power shapes tend to be strongly dependent on operating conditions and fuel management policy and may oscillate spontaneously as a result of xenon feedback.

This paper assesses the significance of space-time problems in reactor design, surveys the analytical and computational tools available for their investigation, reviews pertinent experimental data, describes the new computational tools which are being planned or developed, and indicates potential means of verifying these tools through detailed reactor experiments. The discussion will classify and relate the various analytical approaches.

There are three approaches to the problem of xenon stability. The first

is the "stability in the small" approach in which one examines the linearlized equations which describe the system in the close vicinity of an equilibrium point and attempts to derive criteria for stability of this point. In the Second approach one examines large regions of the state space, possibly with methods of the Lyapunov type, in a search for limit cycles, regions of asymptotic stability, etc. Thirdly, one may attempt to compute a xenon transient in detail, observing its growth or decay. This may be done by the "brute force" method of using a diffusion-depletion code with short time steps or alternately, by using an approximate representation of the spatial dependence (as described below) followed by analog or digital treatment of the time dependence.

We may consider the calculation of general space-time transients as consisting of three segments which proceed simultaneously. The first is the solution of the neutron equations to determine the instantaneous power distribution. The second is the solution of the heat flow and coolant equations to determine the temperature and density distribution knowing the power distribution. The third segment is the calculation of the nuclear cross section, i.e. the coefficients in the nuclear equations, from the instantaneous temperature and density.

Performance of these calculations with a detailed three dimensional spatial description is out of the question at the present time and so most interest has centered on methods for approximating the spatial dependence. Here we distinguish four methods: (i) adiabatic method; (ii) higher order difference or "cell" method; (iii) nodal analysis; and (iv) modal analysis. These will be described and illustrated with sample calculations.